NON-PUBLIC?: N
ACCESSION #: 8901110062
LICENSEE EVENT REPORT (LER)

FACILITY NAME: WATERFORD STEAM ELECTRIC STATION UNIT 3 PAGE: 1 OF 5

DOCKET NUMBER: 05000382

TITLE: REACTOR TRIP RESULTING FROM INADEQUATE ADMINISTRATIVE CONTROL OF WORK
AROUND SENSITIVE EQUIPMENT
EVENT DATE: 12/08/88 LER #: 88-033-01 REPORT DATE: 01/09/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: R. S. Starkey, Operations Superintendent TELEPHONE: (504) 464-3178

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE TO NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 1311 hours on December 8, 1988, Waterford Steam Electric Station Unit 3 was operating at 100% power when a transient induced by the cycling of Power Distribution Panel (PDP) 3014AB breakers caused the reactor to trip on low Departure from Nucleate Boiling Ratio (DNBR). The PDP cover slipped while being removed for maintenance, causing one-half of the PDP breakers to open. The maintenance personnel closed the breakers causing pressurizer pressure instrument control loops to reenergize; this appeared to the Steam Bypass Control System and Reactor Power Cutback System (RPCS) as a large load rejection. Steam bypass control valves quick-opened and the RPCS actuated. Due to the loss of power to Main Turbine (MT) control circuits, the MT was not setback by the RPCS. With the resulting steam demand greater than reactor power, Reactor Coolant System (RCS) pressure decreased and the Core Protection Calculators tripped the reactor on anticipated low DNBR.

The root cause of this event is inadequate administrative control of work

performed on or around equipment that could cause a plant trip or safety system actuation. A directive is being developed to aid in the planning and approval of high risk tasks. Since safety systems functioned to protect the plant, there was no danger to the health or safety of the public or plant personnel.

END OF ABSTRACT

At 1311 hours on December 8, 1988, Waterford Steam Electric Station Unit 3 was operating at 100% power when a reactor trip occurred. The trip was caused by the closing of power distribution circuit breakers that had been inadvertently opened. Since this event resulted in the actuation of the Core Protection Calculators (CPCs) (EIIS Identifier JC-CPU) and Emergency Feedwater (EFW) System (EIIS Identifier BA), the event is reportable as an Engineered Safety Feature (ESF) actuation.

Maintenance Department electricians were removing the front panel cover from non-safety power distribution panel (PDP) 3014AB (EIIS Identifier EE-PL) during performance of work under Design Change (DC) 3070. DC 3070 was issued in response to NRC Information Notice 85-77 and provides the Emergency Notification System (ENS) with an uninterruptable power supply. The electricians were preparing to connect ENS supply power to circuit 23 in PDP 3014AB. As the internal panel cover was being removed, it slipped and knocked open the odd-numbered breakers (number 1 through 29) on PDP 3014AB. This caused a loss of power to various components including Process Analog Control Cabinets (PACCs) 21, 30 and 31 (EIIS Identifier JL), and the Chemical and Volume Control (CVC) (EIIS Identifier CB) isolation relay circuitry. Other circuits lost were various radiation monitoring (EIIS Identifier IL), communication, information, and spare circuits.

The effect of opening these circuit breakers did not immediately jeopardize plant operation. The loss of power to the CVC isolation relays caused CVC letdown from the Reactor Coolant System (RCS) (EIIS Identifier AB) to isolate, and charging pump (CP) (EIIS Identifier CB-P) suction to switch from the Volume Control Tank (VCT) (EIIS Identifier CB-TK) to the Refueling Water Storage Pool (RWSP) (EIIS Identifier BP-RVR). The loss of power to PACCs 30 and 31 caused a loss of pressurizer pressure and level control circuits (EIIS Identifier IO) resulting in deenergizing the pressurizer heaters (EIIS Identifier AB-PZR-EHTR) and started both the backup CPs. The loss of PACC 21, the Turbine Control Panel, placed the Main Turbine (MT) (EIIS Identifier TA) out of automatic operation and into manual.

The electricians attempted to telephone the control room with no immediate success. The operators were preoccupied with evaluating alarms and lost control loops resulting from the deenergized circuits. Unable to contact the control room, the electricians decided to close the breakers and restore power to the affected components.

The restoration of power to the pressurizer control loops caused a rapid rise in the pressurizer pressure control loop output signal; actual pressurizer pressure was stable at this time. The Steam Bypass Control System (SBCS) (EIIS Identifier JI) interpreted the false rapid rise of pressurizer pressure control loop output signal to be a large load rejection and quick-opened all six Steam Bypass Valves (SBVs) (EIIS Identifier SB-FCV). Due to the magnitude of the computed load rejection, the SBCS also signalled the Reactor Power Cutback System (RPCS) (EIIS Identifier JD) to actuate. The RPCS actuated and dropped regulating Control Element Assembly (CEA) (EIIS Identifier AA-ROD) Groups 5&6 to reduce reactor power to approximately 50% and signaled the MT to setback.

During a reactor power cutback, the MT is designed to be setback at 450% turbine load per minute. The loss of power to PACC 21 caused a loss of power to some of the Digital Electro-Hydraulic (DEH) control circuits (EIIS Identifier JJ) which control MT Governor Valve (EIIS Identifier TA-FCV) position. As a result, the MT shifted to manual operation. In manual operation the setback function is lost; however, a turbine runback at 200% turbine load per minute will occur.

The rapid drop of reactor power caused by the cutback combined with the high steam demand from the open SBVs and the turbine runback vice setback, resulted in steam demand being greater than reactor power. RCS temperature and pressurizer (EIIS Identifier AB-PZR) level dropped, causing RCS pressure to decrease. The Core Protection Calculators (CPCs) (EIIS Identifier JC-CPU) anticipated that the rate at which RCS pressure was dropping could cause a low Departure from Nucleate Boiling Ratio (DNBR) and at 1311 hours the CPCs generated a reactor trip.

After the reactor trip, Steam Generator (SG) (EIIS Identifier AB-SG) levels dropped to 27.4% by narrow range indication. All three Emergency Feedwater (EFW) Pumps (EIIS Identifier BA-P) started and EFW isolation valves (EIIS Identifier BA-ISV) 228A&B and 229A&B opened as designed. EFW will not be supplied to the SGs until EFW flow control valves (EIIS Identifier BA-FCV) 223A&B or 224A&B open at 55% or less SG wide range level indication. SG B level was increasing and was attributed to leakage past Main Feed Water Regulating Valve FW173B (EIIS Identifier SJ-FCV). When FW173B was isolated, SG B level appeared to stabilize. The FW173B valve positioner was discovered to be out of calibration and was not completely shutting FW173B on a reactor trip signal. The valve was calibrated and returned to service on December 9, 1988.

At 1319 hours the uncomplicated reactor trip recovery procedure was entered and the plant was placed in hot standby. The post trip review concluded the reactor trip was due to low level in SG A. Also the initial review determined the EFW system actuated but that no flow was introduced into either SG. This information was reported to the NRC during the event notification at 1522 hours on December 8, 1988.

An in-depth investigation of the event was performed and the cause of the trip was determined to be due to low DNBR. The sequence of events was determined to be as presented in this report. Also discovered in the investigation was that EFW223B opened three seconds after the trip, although SG B level had not dropped to 55% wide range level. The water introduced into SG B through EFW223B would not be significant compared to the leakage past FWI73B. EFW223B was also discovered to have not shut following the event. A faulty circuit card was found and replaced to restore EFW223B to normal operation on December 11, 1988. The abnormal valve operation was in the conservative direction and did not threaten plant safety. The investigation also discovered one of the computer printouts of the sequence of events did not register a low DNBR trip. This helped lead the initial review to conclude that the trip was due to low SG levels. A loose connector in the Plant Monitoring Computer (EIIS Identifier IQ-CPU) is attributed to causing the computer error, and the connector has been replaced under WA 01020458.

The root cause of this event is inadequate administrative control of work performed on or around equipment that could cause a plant trip or safety system actuation. Contributing causes of this event were inadequate design of PDP covers and maintenance personnel operating equipment without understanding how the plant could be affected. An Operations & Maintenance Directive is being developed to address work on or around equipment that has a high risk for causing a plant trip or a safety system actuation. Although there is guidance for work on sensitive equipment, the new directive will expand the scope and direction for work which has the potential to cause a plant trip or ESF actuation. This will help personnel in the planning and approval process to identify high risk jobs and ensure appropriate precautions are taken. Station Modification (SM) 1816 had addressed the problem of handling awkward PDP covers by attaching handles to the covers. However, SM 1816 was voided on June 18, 1987, since no practical reason for the modification could be seen at the time. Problem Evaluation/Information Request (PEIR) 60828 was issued in February 1988 to address the problem of handling PDP covers. Design Engineering will respond to PEIR 60828 by January 31, 1989. This event was discussed with site personnel at the December 21 & 22, 1988 Safety Meetings. Maintenance personnel will be further reminded of appropriate action to be taken when operating equipment is affected during maintenance.

Since safety systems functioned to protect the plant during this event there was no threat to the health or safety of the public or plant personnel.

SIMILAR EVENTS

NONE

PLANT CONTACT

R. S. Starkey, Operations Superintendent, 504/464-3178

ATTACHMENT 1 TO 8901110062 PAGE 1 OF 1

Ref: 10CFR50.73(a)(2)(iv)

LOUISIANA
POWER & LIGHT / WATERFORD 3 SES P.O. BOX B KILLONA,
LA 70066-0751

January 9, 1989

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555

SUBJECT: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Reporting of Licensee Event Report

Attached is Licensee Event Report Number LER-88-033-00 for Waterford Steam Electric Station Unit 3. This Licensee Event Report is submitted pursuant to 10CFR50.73(a)(2)(iv).

Very truly yours,

N.S. Carns
Plant Manager - Nuclear

NSC/WMC:rk

Attachment

cc: R.D. Martin, NRC Resident Inspectors Office, INPO Records Center
(J.T. Wheelock), E.L. Blake, W.M. Stevenson, D.L. Wigginton

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